December 19, 2008

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555-0001

10 CFR 50.73

Gentlemen:

TENNESSEE VALLEY AUTHORITY - SEQUOYAH NUCLEAR PLANT (SQN) UNIT 2 - DOCKET NO. 50-328 - FACILITY OPERATING LICENSE DPR-79 - LICENSEE EVENT REPORT (LER) 50-328/2008-001-00

The enclosed LER provides details concerning a manual reactor trip and automatic engineered safety feature (ESF) actuation of auxiliary feedwater. The manual trip occurred as a result of partial loss of main feedwater flow to a steam generator that resulted from a failure of a main feedwater regulating valve controller. This report is being submitted in accordance with 10 CFR 50.73 (a) (2) (iv) (A), as an event that resulted in a valid actuation of the reactor protection system.

Sincerely,

Original signed by

Timothy P. Cleary
Site Vice President

Enclosure
cc: See page 2

Enclosure
cc (Enclosure):

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NRC FORM 366 (9-2007) — U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB NO. 3150-0104 EXPIRES 08/31/2010

Estimated burden per response to comply with this mandatory collection request: 80 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the Records and FOIA/Privacy Service Branch (T-5 F52), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to infocollects@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202, (3150-0104), Office of Management and Budget, Washington, DC 20503. If a means used to impose an information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
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4. TITLE: Manual Reactor Trip Following Partial Loss of Main Feedwater Flow to Loop 4 Steam Generator

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO.	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
11	03	2008	2008	- 001 -	00	12	19	2008	FACILITY NAME	DOCKET NUMBER

9. OPERATING MODE: 1

10. POWER LEVEL: 100

11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: *(Check all that apply)*

☐ 20.2201(b)	☐ 20.2203(a)(3)(i)	☐ 50.73(a)(2)(i)(C)	☐ 50.73(a)(2)(vii)
☐ 20.2201(d)	☐ 20.2203(a)(3)(ii)	☐ 50.73(a)(2)(ii)(A)	☐ 50.73(a)(2)(viii)(A)
☐ 20.2203(a)(1)	☐ 20.2203(a)(4)	☐ 50.73(a)(2)(ii)(B)	☐ 50.73(a)(2)(viii)(B)
☐ 20.2203(a)(2)(i)	☐ 50.36(c)(1)(i)(A)	☐ 50.73(a)(2)(iii)	☐ 50.73(a)(2)(ix)(A)
☐ 20.2203(a)(2)(ii)	☐ 50.36(c)(1)(ii)(A)	☒ 50.73(a)(2)(iv)(A)	☐ 50.73(a)(2)(x)
☐ 20.2203(a)(2)(iii)	☐ 50.36(c)(2)	☐ 50.73(a)(2)(v)(A)	☐ 73.71(a)(4)
☐ 20.2203(a)(2)(iv)	☐ 50.46(a)(3)(ii)	☐ 50.73(a)(2)(v)(B)	☐ 73.71(a)(5)
☐ 20.2203(a)(2)(v)	☐ 50.73(a)(2)(i)(A)	☐ 50.73(a)(2)(v)(C)	☐ OTHER
☐ 20.2203(a)(2)(vi)	☐ 50.73(a)(2)(i)(B)	☐ 50.73(a)(2)(v)(D)	Specify in Abstract below or in NRC Form 366A

12. LICENSEE CONTACT FOR THIS LER

NAME	TELEPHONE NUMBER (Include Area Code)
Zachary T Kitts, Licensing Engineer	423-843-7018

13. COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX
X	SJ	FIC	F180	Y	X	AB	TBG	M270	Y

14. SUPPLEMENTAL REPORT EXPECTED		15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
☐ YES *(If yes, complete 15. EXPECTED SUBMISSION DATE)*	☒ NO				

ABSTRACT *(Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines)*

On November 3, 2008, at 2322 Eastern standard time with Unit 2 operating at 100 percent power, a manual reactor trip was initiated because of a partial loss of main feedwater flow to the steam generator Loop 4. The immediate cause was failure of the Loop 4 feedwater regulating valve controller. During normal power operations, operators received a steam generator level high-low deviation annunciation. An operator identified a decreasing level on the steam generator Loop 4. The controller was placed to manual, but the controller failed to respond to the operator's attempt to increase level. Operators took action to manually trip the reactor. Following the reactor trip, an RCS leak occurred. The leak was from an instrumentation sensing line to the pressurizer. The plant systems responded as designed. The K1 Relay to the Unit 2 Loop 4 main feedwater regulating valve flow indicating controller has been determined as the most probable root cause of this event. The relay failure is attributed to a failing contact connection, which resulted in a slow closing drift of the main feedwater regulating valve.

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17. NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)*

I. PLANT CONDITION(S)

Unit 2 was in Mode 1 operating at approximately 100 percent reactor power.

II. DESCRIPTION OF EVENT

A. Event:

On November 3, 2008, at 2322 Eastern standard time (EST) with Unit 2 operating at 100 percent reactor power, the reactor was manually tripped as a result of continued steam generator (SG) [EIIS Code AB] level decline in Loop 4. Following the reactor trip, the plant systems responded as designed. The rods fully inserted as required. Immediate operator actions were performed, as required. The unit entered Mode 3 and an event investigation was initiated. A reactor coolant system (RCS) leak occurred in a pressurizer instrument sensing line following the reactor trip.

B. Inoperable Structures, Components, or Systems that Contributed to the Event:

None.

C. Dates and Approximate Times of Major Occurrences:

November 3, 2008 at 2320 EST	The SG level high-low deviation alarm annunciated in the main control room (MCR).
November 3, 2008 at approximately 2320 EST	MCR operator monitoring the control boards identified a decreasing SG level in Loop 4.
November 3, 2008 at approximately 2321 EST	The operator placed the feedwater regulator valve controller in manual control. The operator attempted to manually change feedwater flow but, there was no observed change in the feedwater flow.
November 3, 2008 at 2322 EST	Manual actuation of the reactor protection system was initiated because of failure of the feedwater regulator valve controller to respond to manual input and imminent loss of SG level of Loop 4.
November 3, 2008 at approximately 2341 EST	Operators had indications of an unidentified RCS leak (approximately 1-2 gpm) and enter into appropriate abnormal operating procedure. Subsequently, the plant was cooled down and the leak was isolated.

17. NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)*

D. Other Systems or Secondary Functions Affected:

No other systems or secondary functions were affected by this event.

E. Method of Discovery:

MCR operator noticed a decline in Loop 4 SG level.

F. Operator Actions:

MCR operator placed the Loop 4 SG feedwater regulating valve controller [EIIS Code JB] in manual. The operator tried to manually increase the feedwater flow and observed no increase. Manual actuation of the reactor protection system [EIIS Code JC] was initiated as a result of imminent loss of Loop 4 SG level. Control room personnel maintained the unit stable in hot standby, Mode 3.

G. Safety System Responses:

The plant responded to the reactor trip as designed, with one exception. A SG blowdown sample line isolation valve [EIIS Code WI] failed to close as required by the auxiliary feedwater initiation signal. The valve was removed for repair and testing. The valve was reinstalled following satisfactory test results.

III. CAUSE OF THE EVENT

A. Immediate Cause:

The immediate cause was the failure of a Loop 4 SG feedwater regulating valve controller.

B. Root Cause:

The K1 relay associated with the Unit 2 Loop 4 main feedwater regulating valve flow indicating controller has been determined as the most probable cause of this event. The relay failure is attributed to a failing contact connection, which resulted in a slow, closing drift of the main feedwater regulating valve.

C. Contributing Factor:

There were no contributing factors.

IV. ANALYSIS OF THE EVENT

The plant systems responded to the manual reactor trip as designed. The RCS average temperature was near 578.2 degrees Fahrenheit prior to the loss of main feedwater. The pressurizer level was at approximately 60 percent and on program with RCS temperature. Following the reactor trip, the loss of nuclear heat generation resulted in a rapid decrease in RCS average temperature to 548 degrees Fahrenheit, followed by a slower decline to 538 degrees Fahrenheit. The pressurizer level responded to the RCS temperature as designed, although one pressurizer level indicator and pressurizer pressure indicator produced erroneous values. The response of the transmitter was a result of the failed instrument sensing line that resulted in an unidentified leakage rate of approximately 1-2 gpm. As heat removal in the steam generators decreased, the decrease in RCS temperature slowed. The introduction of cold auxiliary feedwater (AFW) resulted in a slower, but continued reduction in RCS temperature until AFW flow was reduced after approximately 10 minutes following the reactor trip. RCS temperature then started to increase. RCS temperature remained within technical specification limits and bounded by the Safety Analysis Report (SAR) analysis.

The plant responded as expected for the conditions of the trip. No technical specification limits were exceeded and the SAR analysis of this event remained bounding.

V. ASSESSMENT OF SAFETY CONSEQUENCES

Based on the above "Analysis of The Event," this event did not adversely affect the health and safety of plant personnel or the general public.

VI. CORRECTIVE ACTIONS

A. Immediate Corrective Actions:

Control Room personnel responded as prescribed by emergency procedures. They diagnosed the plant condition and took necessary action to stabilize the unit in a safe condition. In addition, Control Room personnel used abnormal operating procedures to diagnose the source of the RCS leak. Plant personnel identified the leak in a pressurizer instrument sensing line and terminated the leak once safe access to the pressurizer enclosure was achievable. The instrument sensing line (i.e., the flexible bellows portion) was replaced.

B. Corrective Actions to Prevent Recurrence:

The feedwater control system is scheduled to be upgraded during the Unit 2 Cycle 16 and Unit 1 Cycle 17 refueling outages.

17. NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)*

Because of the industry events documenting problems with the K1 relay that results in the relatively slow, subtle controller output degradation, it has been determined that the cycling of these controllers promotes a cleaning action on the contacts to help ensure proper operation when taken to manual. The preventative maintenance frequency is being increased to cycle the auto/manual switches on a more frequent basis.

VII. ADDITIONAL INFORMATION

A. Failed Components:

A SG blowdown sample flow solenoid valve [Target Rock, Model No. 82KK-002] failed to function as required during the event. The valve was removed, cleaned, and stroke tested with satisfactory results and reinstalled.

A pressurizer instrumentation flexible sensing line [Parker Metal Bellows, Model No. 73989] developed a leak. The flexible bellows sensing line was replaced. An inspection of the other pressurizer instrumentation flexible sensing lines found no evidence of damage due to improper bend radius.

B. Previous LERs on Similar Events:

A review of previous reportable events identified SQN LER 50-328/2007-002-00. In March 13, 2007, with SQN Unit 2 operating at 100 percent power, operating personnel initiated a manual reactor trip because of loss of main feedwater flow to each of the steam generators. The plant systems responded as designed. The cause of the flow loss was a failure of the 2A main feedwater pump control system. The root cause of the event was determined to be a faulty local/remote switch which is internal to the speed indicating controller. A K1 relay, which is internal to the main feedwater pump speed indicating controller, was found to be faulty and as a result determined to be a contributing cause.

C. Additional Information:

None.

D. Safety System Functional Failure:

This event did not result in a safety system functional failure in accordance with 10 CFR 50.73(a)(2)(v).

E. Unplanned Scram with Complications:

This condition did not result in an unplanned scram with complications.

VIII. COMMITMENTS

None.